

**Virginia Electric and Power Company
Surry Power Station
5570 Hog Island Road
Surry, Virginia 23883**

March 19, 1997

U. S. Nuclear Regulatory Commission
Document Control Desk
Washington, D. C. 20555

Serial No.: 97-117
SPS:BAG
Docket No.: 50-281
License No.: DPR-37

Dear Sirs:

Pursuant to Surry Power Station Technical Specifications, Virginia Electric and Power Company hereby submits the following Licensee Event Report applicable to Surry Power Station Unit 2.

REPORT NUMBER

50-281/97-001-00

This report has been reviewed by the Station Nuclear Safety and Operating Committee and will be forwarded to the Management Safety Review Committee for its review.

Very truly yours,



D. A. Christian
Station Manager

Enclosure

Commitments contained in this letter: None

copy: Regional Administrator
101 Marietta Street, NW, Suite 2900
Atlanta, Georgia 30323

R. A. Musser
NRC Senior Resident Inspector
Surry Power Station

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PDR ADDCK 05000281
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CATEGORY 1

REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR:9703260311 DOC.DATE: 97/03/19 NOTARIZED: NO DOCKET #
 FACIL:50-281 Surry Power Station, Unit 2, Virginia Electric & Powe 05000281
 AUTH.NAME AUTHOR AFFILIATION
 CHRISTIAN,D.A. Virginia Power (Virginia Electric & Power Co.)
 RECIP.NAME RECIPIENT AFFILIATION

SUBJECT: LER 97-001-00:on 970218,manual reactor trip & ESF actuation
 occurred due to loss of EHC control power.Caused by
 momentary short.Relay card was replaced.W/970319 ltr.

DISTRIBUTION CODE: IE22T COPIES RECEIVED:LTR 1 ENCL 1 SIZE: 6
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LICENSEE EVENT REPORT QUALITY ASSURANCE CHECK SHEET

LER NUMBER: 28197001

BATCH: 1077

STUDY: ES

PAGES: 5

TITLE: MANUAL REACTOR TRIP AND ESF ACTUATION DUE TO LOSS OF EHC
CONTROL POWER

EVENT DATE: 02/18/97

LER REVISION: 00

OTHER FACILITIES:

OPERATING MODE: N

APPLICABLE CFR: M

POWER LEVEL: 100

AUTHOR:

CHRISTIAN, D. A.

NPRDS REPORTABILITY

CAUSE

X

SYSTEM

TG

COMPONENT

RLY

MANUFACTURER

W120

NPRDS

N

CONTINUED:

SUPPLEMENT: N

SUPPLEMENT DATE:

QA BY:

QA DATE:

12

2/15/97

1077
LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS MANDATORY INFORMATION COLLECTION REQUEST: 50.0 HRS. REPORTED LESSONS LEARNED ARE INCORPORATED INTO THE LICENSING PROCESS AND FED BACK TO INDUSTRY. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (T-6 #33), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

FACILITY NAME (1)

SURREY POWER STATION, Unit 2

DOCKET NUMBER (2)

05000 - 281

PAGE (3)

1 OF 5

TITLE (4)

Manual Reactor Trip and ESF Actuation Due to Loss of EHC Control Power

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCUMENT NUMBER
02	18	97	97	-- 001 --	00	03	19	97	FACILITY NAME	05000-
OPERATING MODE (9)		N	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)							
POWER LEVEL (10)		100 %	20.2201(b)		20.2203(a)(2)(v)		50.73(a)(2)(i)		50.73(a)(2)(viii)	
			20.2203(a)(1)		20.2203(a)(3)(i)		50.73(a)(2)(ii)		50.73(a)(2)(x)	
			20.2203(a)(2)(i)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)		73.71	
			20.2203(a)(2)(ii)		20.2203(a)(4)		x 50.73(a)(2)(iv)		OTHER	
			20.2203(a)(2)(iii)		50.36(c)(1)		50.73(a)(2)(v)		Specify in Abstract below	
			20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)		or in NRC Form 366A	

LICENSEE CONTACT FOR THIS LER (12)

NAME	D. A. Christian, Station Manager	TELEPHONE NUMBER (Include Area Code)	(757) 365-2000
------	----------------------------------	--------------------------------------	----------------

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS
X	TG	RLY	W120	No					

SUPPLEMENTAL REPORT EXPECTED (14)

YES	X	NO	EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
(If yes, complete EXPECTED SUBMISSION DATE).						

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On February 18, 1997, at 1449 hours, with Unit 2 operating at 100% reactor power, a loss of turbine Electro-Hydraulic Control (EHC) control power caused the turbine governor valves to close. The reactor was manually tripped at 1450 hours. The Reactor Protection System actuated and functioned as designed, and all control rods inserted into the core. Three Individual Rod Position Indicators (IRPI) were observed between 10 and 15 steps. The Turbine Driven Auxiliary Feedwater Pump (TDAFWP) started and operated as designed, however, a second start was initiated and the pump tripped on overspeed.

The most probable cause of the loss of EHC control power was a momentary short that lead to an over-voltage condition and tripped two redundant +15 VDC power supplies. An automatic turbine trip should have been initiated by the loss of EHC control power, but did not occur. The lack of a turbine trip was due to a faulty relay card. The power supplies and relay card were replaced. Two IRPIs were recalibrated and the calibration on the remaining IRPI verified with recent data. Emergency procedures were revised to reduce the possibility of a re-start signal until the TDAFWP governor reset. No conditions adverse to safety resulted from this event and the health and safety of the public were not affected. This event is being reported pursuant to 10 CFR 50.73(a)(2)(iv).

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TEXT CONTINUATION

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SURRY POWER STATION, Unit 2	05000 - 281	97	-- 001 --	00	2 OF 5

TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

1.0 DESCRIPTION OF THE EVENT

On February 18, 1997, Unit 2 was operating at 100% reactor power when at 1449 hours, two redundant +15 VDC power supplies [EIS-TG,JX] in the Turbine Electro-Hydraulic Control (EHC) [EIS-TG] cabinet tripped. This resulted in a loss of several turbine control panel indications and caused several plant computer [EIS-ID] alarms relating to turbine controls. The turbine governor valves moved in a closed direction due to the loss of EHC control power. The Operator responded to the loss of electrical load and reactor coolant system (RCS) temperature increase by manually tripping the reactor and turbine at 1450 hours. At the time of the trip, reactor power was at 90% and the turbine generator output was at approximately 650 MWe.

The Reactor Protection System (RPS) [EIS-JC] actuated and functioned as designed, and all control rods inserted into the core. The Nuclear Instrumentation responded as expected and the primary RCS temperature decreased to a stable no-load condition following the trip. No primary safety or power operated relief valves were actuated during the event. No secondary safety relief valves or power operated relief valves actuated during the transient. All electrical busses transferred properly following the trip and the emergency diesel generators were operable. There were no radiation releases due to this event, nor were there any personnel injuries or contamination events.

The loss of the two redundant +15 VDC power supplies should have resulted in an automatic turbine trip signal due to the loss of EHC control power, however, turbine trip did not occur.

All control rods inserted into the core as indicated by the rod bottom light indication. Three Individual Rod Position Indicators (IRPIs) [EIS-AA,ZI], J13, F12, and F6, were observed to initially indicate between 10 and 15 steps. IRPIs J13 and F12 drifted to zero steps approximately 5 minutes after the trip. IRPI F12 indicated less than 10 steps 96 minutes after the trip. In response to the IRPI indication above 10 steps, additional boron was added to the RCS using the emergency boration flowpaths. The shutdown margin for Unit 2 was verified to be satisfactory.

All three auxiliary feedwater pumps automatically started as designed on low-low Steam Generator (SG) levels following the trip. Subsequent to the trip, an auxiliary operator found the Turbine Driven Auxiliary Feedwater Pump (TDAFWP) [EIS-BA,P] Trip Throttle Valve [EIS-BA,SHV] in a tripped position. Review of the post-trip data indicated that the TDAFWP started and ran normally for approximately 17 minutes and then shutdown normally after the Anticipated Transient Without Scram Mitigation System Actuation Circuitry (AMSAC) was reset and the low-low SG level actuation signals cleared. Within a few seconds of the TDAFWP steam supply valves closing, the valves received a second open signal. The TDAFWP restarted and then tripped on overspeed.

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TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

A 4-hour Non-Emergency report to the NRC Operations Center was made in accordance with 10 CFR 50.72(b)(2)(ii) due to a manual or automatic engineered safety feature (ESF) actuation. This report is being made pursuant to 10 CFR 50.73(a)(2)(iv), any event or condition that resulted in manual or automatic actuation of any ESF.

2.0 SAFETY CONSEQUENCES AND IMPLICATIONS

In response to the loss of electrical load and increase to the RCS temperature, the Operator took appropriate actions and manually tripped the reactor and turbine. Upon receipt of the reactor trip signal, the RPS actuated and functioned as designed. All control rods inserted into the core as indicated by the rod bottom light indication. All three auxiliary feedwater pumps automatically started on low-low SG levels, as designed. The Motor Driven Auxiliary Feedwater Pumps [EHS-BA,P] continued to provide residual heat removal following the TDAFWP trip on overspeed. Plant response was as expected and the unit stabilized at hot shutdown. No conditions adverse to safety resulted from this event and the health and safety of the public were not affected.

3.0 CAUSE

The direct cause of the manual reactor trip and the subsequent initiation of auxiliary feedwater was the loss of the two redundant +15 VDC power supplies for the Turbine EHC Control System. Detailed testing of the power supplies and inspection for degraded subcomponents was conducted with no abnormalities noted. The most probable root cause of the power supply failure was a momentary short on the +15 VDC distribution bus that lead to an over-voltage condition on the +15 VDC power supplies.

Prior to the trip, at approximately 1445 hours, Instrument and Control (I&C) personnel started preventive maintenance (PM) activities associated with cleaning the cooling fan and filter in the Unit 2 Turbine EHC cabinet. The EHC cabinet doors had been opened, the cabinet cooling fan disconnected, and the filter removed. Technicians then left the cabinet area to clean the removed filter assembly. As part of the subsequent troubleshooting activities, the PM activity performed prior to the trip was repeated. During the troubleshooting, the power supplies remained stable. The troubleshooting activity did not conclude that the PM activity was the cause for the loss of the power supplies.

The EHC Relay Card at location 1A02Q1 [EHS-TG,RLY] is designed to generate a turbine trip signal on the loss of EHC control power. Failure analysis on this card indicated an internal failure of two relay assemblies. Both relay assemblies had intermittent internal failures. The turbine trip signal due to the loss of EHC control power is part of the turbine protection logic but does not provide input into the RPS.

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The overspeed trip of the TDAFWP was due to the small tolerance between the SG levels that initiate and clear the TDAFWP start signal. Within a few seconds of the TDAFWP Steam Supply valves closing, the valves received a second open signal when Steam Generator A decreased to less than 17%, completing the logic for the start of the TDAFWP. When the turbine attempted to restart, the turbine governor had not yet reached its low speed stop reset pressure. This resulted in the overspeed trip condition on the turbine and the tripping of the turbine trip throttle valve. Discussions with the vendor indicated that an inherent time period exists with the Woodward PG-PL governor during which, if the turbine restarts, the governor will not reset to the low speed limit, resulting in an overspeed condition on the turbine. This time period ranges from 9 to 16 seconds.

4.0 IMMEDIATE CORRECTIVE ACTIONS

Following the trip, Control Room Operators acted promptly to place the plant in a safe hot shutdown condition in accordance with emergency and other operating procedures.

The Shift Technical Advisor calculated the shutdown margin and monitored the critical safety function status trees to verify that the unit conditions were acceptable. Plant response was as expected and the unit was stabilized at hot shutdown.

5.0 ADDITIONAL CORRECTIVE ACTIONS

A Root Cause Evaluation (RCE) team was established to determine the cause of the event and to develop recommendations to prevent recurrence. The team reviewed the troubleshooting performed on the EHC cabinet and power supplies, but did not determine the cause of the power supply over-voltage condition. Detailed testing and inspection of the power supplies for degraded subcomponents were conducted with no abnormalities noted. The +15 VDC power supplies were replaced.

A failure analysis was conducted on the Relay Card at location 1A02Q1 and the results indicated internal failure on two relay assemblies. Both relay assemblies had intermittent internal failures. The relay card was replaced.

To provide increased assurance that the TDAFWP will not overspeed prior to the governor being able to reset to the low speed stop, Emergency Operating Procedures were revised to provide a larger margin between the auto-start of the pump and the SG level at which the pump is secured.

Three IRPIs indicated between 10 and 15 steps initially, however, two of IRPIs drifted to zero steps approximately 5 minutes after the trip. These two IRPIs were recalibrated. The third IRPI indicated less than 10 steps 96 minutes after the trip. The calibration on the third

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TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

IRPI was verified with recent calibration data and the IRPI was returned to service. These same three IRPIs failed to indicate less than 10 steps after an August 3, 1996 reactor trip. IRPI F6 also failed to indicate zero after a reactor trip on December 13, 1996. The failure of these IRPIs to indicate less than 10 steps after a reactor trip is being addressed by the Corrective Action Program.

As part of the RCE, equipment problems were evaluated in accordance with the Maintenance Rule. It was determined that a Maintenance Rule Functional Failure had occurred because of the manual reactor trip. A Maintenance Preventable Functional Failure did not occur since no failed components that lead to the trip were discovered during the troubleshooting.

6.0 ACTIONS TO PREVENT RECURRENCE

A detailed Reactor Trip Report and an Root Cause Evaluation is being performed for this event. Additional approved recommendations from the Root Cause Evaluation will be implemented in accordance with the Corrective Action Program.

7.0 SIMILAR EVENTS

LER 1-92-001, Dropped Rod Due to Personnel Error Followed By A Required Manual Reactor Trip. The primary +15 VDC power supply had failed low resulting in a loss of EHC control power. Degraded capacitors were discovered in the power supply unit. The power supplies were replaced.

8.0 MANUFACTURER/MODEL #

Manufacturer: Westinghouse Relay Card (EHC)
Equipment: Assembly # 398737, Revision C, Serial # 95805

9.0 ADDITIONAL INFORMATION

Unit 1 was operating at 100% reactor power.